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Department of Energy Technology Annual Progress Report 1 January - 31 December 1982

Risø National Laboratory, DK-4000 Roskilde, Denmark
April 1983

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DEPARTMENT OF ENERGY TECHNOLOGY
ANNUAL PROGRESS REPORT
1 January - 31 December 1982

Abstract. The general development of the Department of Energy Technology at Risø during 1982 is presented, and the activities within the major subject fields are described in some detail. Lists of staff, publications, and computer programs are included.

INIS/EDB descriptors: HEAT TRANSFER; REACTOR PHYSICS; REACTOR TECHNOLOGY; RELIABILITY; RESERVOIR ENGINEERING; RESEARCH PROGRAMS; RISOE NATIONAL LABORATORY

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1. DEVELOPMENT DURING 1982

1.1. The Department of Energy Technology

The change in interest from nuclear energy technology towards energy technology problems in general, at Risø and in the Department, was made manifest by the change of name of the Department to "The Department of Energy Technology", on 1 January 1982.

At the same time the work within the reservoir modelling group was strengthened and it was made a separate entity within the Department.

The work on fluidized-bed combustion of coal was initiated during 1981, and in the beginning of 1982 it was accepted as an energy project with funds from the Energy Research Programme of the Ministry of Energy. An experimental 200-kW fluidized bed was constructed during the year and started up late in the year, with the purpose of studying the combustion process and testing instruments.

Similarly a project of modelling environmental effects of energy production was taken up when it was allocated funds from the Energy Research Programme.

The Reactor Physics Section had some success in its efforts to gain contacts and contracts, and at the same time a collaboration on severe accident analysis began with the Danish utilities, resulting in an increased effort in nuclear technology.

The Nordic collaboration on probabilistic risk analysis, small break analysis, and heat transfer correlations, is progressing well, and several demanding benchmark studies, within the Nordic group as well as within the EEC, have been pursued.

1.2. System and Reliability Analysis

The main part of the work is concerned with developing methods for assessing the reliability of systems and components in nuclear power plants and other industrial installations. Furthermore, a core simulator is being developed, and a limited effort is being spent on monitoring the general trends within the nuclear fields.

The competence regarding reliability analysis, which originates from the nuclear fields, is used for safety and risk analysis of industrial installations. Accordingly, for the past several years the Section has been engaged in analyses of that kind. These include offshore installations, chlorine production facilities, and the Danish natural gas system. During 1982 the main project was an analysis of the security of supply for the proposed natural gas system in the southern part of Sweden. Furthermore, the safety of the Als Sund gas transmission line crossing in Denmark was analysed.

The main effort of the section is given to the 4-year Nordic cooperation: "Probabilistic Risk Analysis and Licensing". During 1982 data problems and benchmark studies for the verification of methods and computer codes were studied.

The reliability benchmark set up within the EEC concerning the emergency feedwater system in a 1300-MWe French nuclear power station is nearly completed.

The core simulator program COSMA combines a module for calculating the reactor physics of the power history with a module for predicting fuel failure. This work is done in collaboration with the Section of Reactor Physics and Dynamics and with financial support from the Danish utilities. The analysis of local pin power was taken up in a Ph.D. study which has been completed.

1.3. Reactor Physics and Dynamics

As in previous years the work in the Section has been concentrated on verifying of already existing models for core follow studies, reactor core dynamics and nuclear power plant dynamics. The work has been influenced by the change of the main Risø computer from a Burroughs B6800 to the larger B7800. This gives better possibilities for solving larger reactor physics problems; however, the changeover period has delayed many of the ongoing projects.

The verification work for three-dimensional models for BWR's and PWR's has been continued. Very satisfactory results have now been obtained in calculations of power distributions for a PWR, while testing the dynamics part of the PWR-model ANTI against experiments still remains.

A new three-dimensional model, NEM, for neutronics calculation based on the nodal expansion method is now operational and if it proves successful in testing, it will be incorporated into the existing PWR and BWR models.

The Section has continued its cooperation with the Danish power utilities to use and develop the fuel management program system SOFIE. Also in cooperation with the Danish power utilities, a project has been formulated to develop and verify the core simulator further. This is a program system intended to predict fuel failures. The project has not yet been started, but will begin in 1983.

The Section is presently involved in the development of a compact-simulator to train operational personnel for a power reactor owned by a utility outside Denmark. The compact-simulator will be based on a power-plant model developed at Risø and it will be adapted to a small computer at the power station.

During the last few years some work has been done to start a project to develop a model to assess environmental effects of energy production. During 1982 governmental funding has now been obtained and thus work has now started.

1.4. Heat Tranfer and Hydraulics

The transition from nuclear to non-nuclear work, which began in 1981, has increased during 1982.

Both the experimental and theoretical nuclear work are concentrated on accident analysis, especially loss-of-coolant and emergency core cooling.

Several reflood and quenching experiments have been performed in two test rigs. In one of these rigs real fuel rods are simulated by electrically heated test pins in annular geometry. The main purpose of this experiment is to investigate the influence of the pin composition and heating method on the quench front velocity. In the other test rig natural circulation and quenching during refloodding in parallel channels are studied. The rig contains 3 channels of different sizes and individual heating.

The theoretical work has continued using a series of different advanced computer codes. The NORCOOL II code has been adapted to predict the results from the above-mentioned two reflood test rigs. Small break loss-of-coolant accidents have been examined using the codes RELAP, TRAC, RAMONA, and SMABRE. Severe accidents have been simulated using the codes MARCH and HAARM-S dealing with core meltdown and aerosol transport, respectively.

The main part of the non-nuclear work has dealt with fluidized-bed construction. The goal is to establish a general theoretical and experimental knowledge of coal pressurized fluidized-bed combustion. Existing fluidized-bed plants have been carefully studied to get an impression of the state of the art.

In order to gain practical experience in fluidized-bed technology, a small fluidized-bed test rig working at atmospheric pressure has been constructed, and it was succesfully started late in the year. The main purpose of this rig is to test and develop instruments for use in fluidized-bed systems and to examine the influence of different parameters on the combustion efficiency and emission of SO_2, NO_x, and particles.

Two new non-nuclear projects were started this year: an investigation of blow-down from pressure vessels containing liquids near the boiling temperature and a theoretical and experimental study of heat exchangers.

The Temperature Calibration Laboratory has had a further increase in jobs for customers in private industry.

1.5. Reservoir Models

During 1982 a separate reservoir group was established, mainly as a consequence of the expansion of the project on oil and gas reservoir modelling.

The reason for this is the recognized need for reservoir simulation in connection with the development of the hydrocarbon reservoirs in the Danish part of the North Sea.

The work has been concerned partly with the implementation and testing of black-oil simulators, in fulfilment of contracts awarded by the Danish Ministry of Energy, partly with field studies for two Danish North Sea reservoirs undertaken for the Danish Energy Agency.

The group also participates in the Danish project concerning heat storage in aquifers. The work has consisted of simulations and the development of software for control and data recording purposes.

1.6. Danish Reactor No. 1

On 15 August 1982 the reactor had been in operation for 25 years. As 1 MWd corresponds to 1 gram U-235, and as the total thermal production has been below 10 MWh, less than 1 gram U-235 has been used during this period.

Every second year a core sample has been taken from the reactor in order to follow any corrosion of the core tank. From the content of chromium and nickel present in the core sample, it is estimated that the reactor can be operative for a further 25 years.

An increasing interest in performing experiments at the reactor has been expressed by high school classes participating in reactor courses. A number of students from various universities and technical high schools have carried out routine experiments at the reactor.

The reactor has been used as a neutron source for neutron radiography. A new temperature-regulated etchbath for plastic foils is under construction. To improve the radiographic images, some radiographs have been digitized.

2. ACTIVITIES OF THE DEPARTMENT

2.1. Probabilistic Risk Analysis and Licensing

In 1981 a Nordic co-operative project was initiated in order to study problems related to the possible use of probabilistic risk analysis methods in the licensing of nuclear power plants. The Department participates in this work with an effort of approximately 2 person years per year. The work is sponsored partly through the Nordic Liaison Committee for Atomic Energy (NKA).

The Department's work on the project during 1982 has taken place mainly within two subject areas: data problems and benchmark studies for the verification of methods and computer codes.

An analysis of pipe failures in Swedish nuclear power plants has been carried out and the preliminary results have been reported. The analysis was based on information from reports on safety-related occurrences from the Swedish Nuclear Power Inspectorate, from reports contained in the Swedish ATV data base, and from interviews. The work will be concluded in spring 83.

In the last three months of the year a study of various data treatment methods was initiated. Groups from VTT (Finland), Studsvik (Sweden) and Risø carried out analyses of five data cases. The results and comparisons of the methods were discussed during a two-day workshop in Studsvik in December. The work will be finished before summer 83.

In order to compare methods and codes, groups from VTT, Studsvik, and Risø have simultaneously carried out an unavailability analysis of a simple High-Pressure Injection System (HPIS) for a Pressurized-Water Reactor. Methods were compared for data collection, qualitative failure analysis, and the quantification of the unavailability. Comparison showed among other things that the final result (the unavailability) was more sensitive to the differences in the data sets than to the differences in the calculation methods.

In the latter part of the year a new benchmark study was initiated. The purpose of this study is to investigate sequences leading to automatic depressurization of a Boiling Water Reactor. In the first phase of this study event reports from Swedish reactors have been examined in order to identify potential precursors to automatic depressurization.

In March 1982 a seminar was arranged in Helsingør, where the results obtained so far in the project were presented. Proceedings of the seminar have been issued (Lauridsen et al., 1982).

REFERENCE

LAURIDSEN, K. et al. (1982). Probabilistic Risk Analysis and Licensing. Proceeding of seminar 2, Helsingør, 29-31 March 1982. Risø-M-2363.

2.2. Reliability Benchmark Exercise

On the initiative of the EEC working group 2 - Water Reactor Safety Research - a reliability benchmark exercise (RBE) was started in September 1981, with a secretariat in the JRC at Ispra. The exercise in which 10 foreign companies and institutes participate will be finished in the middle of 1983.

The reliability benchmark exercise comprises an analysis of the probability of failure of the auxiliary feedwater system in the French Paluel nuclear power stations (4 x 1300 MWe) (see Fig. 1). The analysis is a part of a probabilistic risk assessment, and it is carried out assuming a break in the main feedwater line at the junction to steam generator No. 1.

The following objective for the exercise was established:

1. to demonstrate the maturity of the discipline with its advantages and limitations,

2. to assess the degree of consistency among the results obtained by different organizations and for different methods, and

3. to attempt to define common analysis procedures with possible variants.

The exercise is being executed in the three phases:

a) Qualitative analysis
b) Logical modelling
c) Quantitative analysis

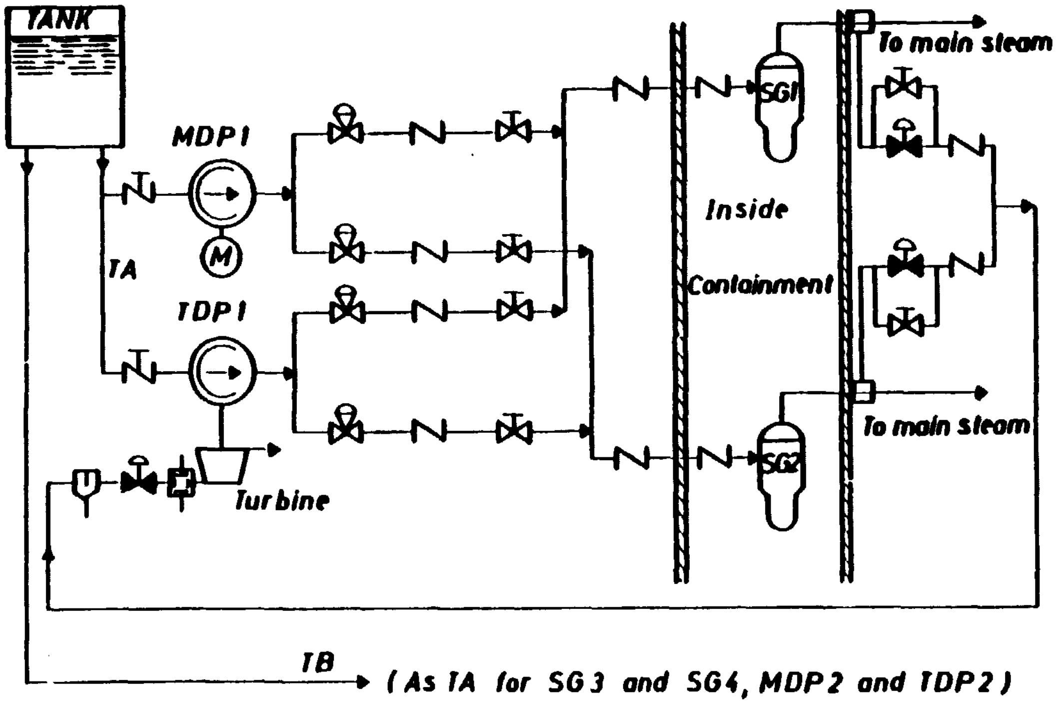


Fig. 1. Simplified scheme of the auxiliary feedwater reference system.

At the time of the turn of the year 1982/83 the qualitative analysis was finished. A report on the qualitative analysis is being prepared, and the logical modelling is in progress.

The qualitative analyses included: A component failure mode and effects analysis, an analysis of the efforts of the failure of interfacing systems, a common cause failure analysis, and a human error analysis. The analyses mentioned were carried out by using common formats for all the participants.

The Department of Energy Technology participates in the reliability benchmark exercise. Our main experience from the work until now is as follows:

A detailed qualitative analysis including a clarification of system performance under the failure conditions considered is a necessity before any probabilistic quantification can be made. Such an analysis should be carried out through a close cooperation with the designer and operator. However, we found that this was not possible in the reliability benchmark exercise to the extent preferred.

A detailed qualitative analysis of a system like the one in question is a very time-consuming task, if done by hand, if a reasonable degree of completeness is to be expected. Thus, some kind of computer assistance seems advisable. Further, a specification of the strategy applied in the search for failures seems a necessity in order to obtain a well-defined analysis and result.

According to the reliability benchmark exercise rules each participant delivered a best estimate of the probability of the "top event" together with the result of the qualitative analysis. These preliminary estimates showed a greater spread than expected which will be clarified in the latest phases of the exercise.

2.3. Core Simulator

For the past several years work has been in progress on the development of a computer code system, COSMA, to predict fuel failures in power reactors. The program system contains modules for cross-section generation, three-dimensional power and burn-up calculations, local pin power determinations, and fuel failure prediction.

In the first version of COSMA (core simulator manager) the pin powers are predicted using a modulation scheme in which the smooth power distribution in a homogenized node is determined by applying an interpolation procedure based on the average nodal powers. The smooth power distributions are multiplied by form functions. The form functions are taken to be the power distribution from the fuel assembly calculation using zero net current boundary conditions.

2.4. Local Pin Power

A study to investigate methods for calculating local pin power has been carried out as a Ph.D. study. The following three most commonly employed principles of calculating local pin powers have been programmed and tested:

1. The Normalization Method where the power distribution from the assembly calculation using zero net current boundary conditions is renormalized to the average values from the global solution,

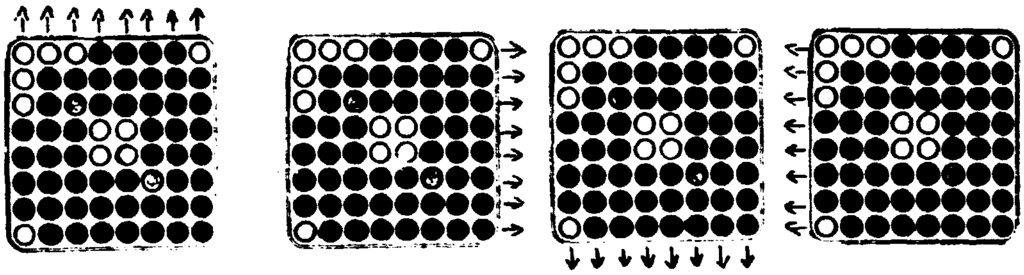
2. the Flux-lupe Method where the boundary parameters from the converged coarse-mesh solution are used as boundary conditions in a number of imbedded assembly calculations, and

3. the Modulation Methods where the smooth power distribution from the coarse-mesh solution is multiplied by some precalculated form functions.

A new local pin power model called the Superpositon Method has been developed at Risø. The basic principle of this method consists of expanding the solution to the heterogeneous assembly problem with boundary conditions derived from the global solution. Precalculated solutions to a number of heterogeneous assembly problems are used as expansion functions. Figure 2 illustrates the principle of the Superposition Method in the case of a flat approximation of the boundary parameters. The Superposition Method has proved to be superior to the other pin power models examined provided both efficiency and accuracy are considered. If the accurate heterogeneous boundary parameters values can be derived from the global solution, one can predict pin powers with a maximum error of about 5% using only six base solutions.

One of the major findings is the close connection between the homogenization scheme, the nodal coarse-mesh method, and the local pin power model.

The base-solutions in a flat approximation



The local pin power determination

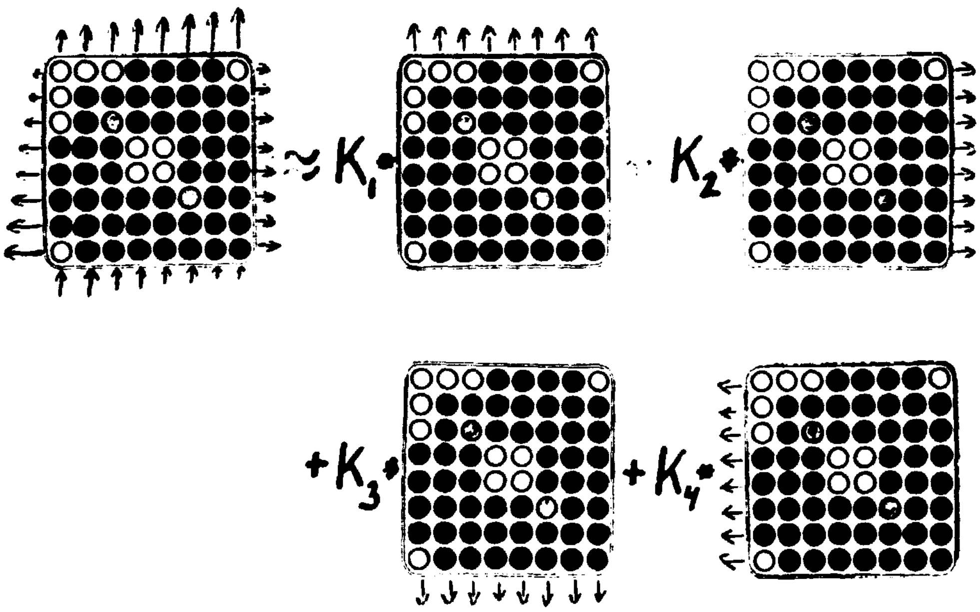


Fig. 2. The principle of the Superposition Method in the case of a flat approximation of the boundary parameter values.

Historically, these three subjects were investigated in sequence, and it is only now after progress has been made in all three areas that the close connection can be recognized. For instance, it has been shown that in practical use of the local pin power models the accuracy is poor if the coarse-mesh solution is based on flux-weighted homogenized parameters. Consequently, other homogenization schemes should be introduced, if the pin powers are to be accurately predicted.

Several new homogenization schemes have been proposed. In two of these schemes: Equivalence Theory (Koebke, 1978) and Generalized Equivalence Theory (Smith, 1980), an additional degree of freedom has been added to the homogenized parameters in order to make them fulfil the requirement of preserving the average currents into the region to be homogenized. The additional degree of freedom lies in postulating flux-discontinuity between adjacent homogenized regions. The discontinuity of the flux is represented by additional homogenization parameters, the discontinuity factors.

One of the major problems concerning these methods is that the region to be homogenized has to be imbedded in this natural environment. Consequently, an iteration procedure between homogenization and coarse-mesh calculation is required in order to obtain fairly accurate solutions.

However, based on the results from the Superposition Method, it proves to be possible to avoid the rather time-consuming iteration procedure using the superposition scheme in the global three-dimensional coarse-mesh calculation. The idea is to update the homogenized parameters during the coarse-mesh calculation using the expansion constants from the local pin power solution scheme, and multiply these constants by the homogenized cross-sections belonging to the precalculated base-solutions. This idea of using the Superposition Method both as a homogenization method and as a means of finding local pin powers has not been examined but it seems to be an interesting topic for further investigations.

REFERENCE

KOEBKE, K. (1978). A New Approach to Homogenization and Group Condensation, IAEA-TECDOC 231, p. 303, IAEA Technical Comm. Mtg Lugano, Switzerland.

SMITH, K.S. (1980). Spatial Homogenization Methods for Light Water Reactor Analysis, Thesis, Mass. Institute of Technology, Cambridge, Mass.

2.5. Study of the Effect of Reduced Enrichment in the Fuel of the DR 3 Research Reactor

The concern of the Carter administration about the safety aspects of the considerable amount of highly enriched (weapons grade) uranium floating around the world for use in research reactors initiated a feasibility study within the framework of IAEA of the consequences of reducing the enrichment from the present 93% to 45% or even 20%. An IAEA guidebook on this subject will be issued shortly. Risø participated in the study in which calculational methods were compared and adjusted, until participants felt confident that they could predict the behaviour of their reactors loaded with lower enriched fuels. As one of the tests of the Risø codes a calculational follow-up of an actual sequence of fuel changes in the DR 3 reactor was performed.

In this excercise tables of diffusion parameters were first prepared by burnup calculations with the CCC programme. (Højerup, 1976). Next, the DBU programme (Lindstrøm Jensen, 1970) interpolates in these tables and performs an overall diffusion calculation and associated calculations of power distribution and burnup distribution.

The DBU programme can be instructed via input data to replace fuel elements at any time by fresh ones. This option was used to follow the actual sequence of changes.

The agreement between the calculated values of U-235 content and experimentally determined ones was quite good. Also, the calculated reactivity increments of the fuel replacements seem reasonable, as the end-of-cycle reactivities did not show any consistent rise or fall during the 9 periods (22 full-power days), where all fuel elements were changed at least once. Figure 3 shows the comparison between calculated and measured U-235 content (grammes of U-235 per fuel element) in the 26 elements. The fresh-element loading is approximately 150 g U-235 in most of the elements and approximately 120 g U-235 in 3 or 4 elements.

	1	2	3	4	5	6
		1	2	3	4	
A		100.6 100.8	130.9 130.2	97.9 96.3	86.4 87.0	
B	86.0 88.4	76.3 74.0	114.4 113.1	127.5 126.0	129.7 129.3	106.8 108.2
C	110.0 109.4	79.9 78.2	73.7 69.5	67.3 63.6	65.5 63.1	60.6 61.6
D	92.4 92.9	117.1 115.8	100.7 98.6	114.6 112.3	85.7 84.0	114.2 112.6
E		76.5 77.0	118.7 118.1	97.0 95.5	79.4 80.6	

Fig. 3. Distribution of U235 at the end of the last calculated reactor period for the 93% case. Comparison is made with DR 3's own estimates based ion flux measurements. Upper numbers are the calculated values, lower ones are the measured values.

Figure 4 shows the calculated k_{eff} as function of time and fuel changes. The solid lines are 93% enriched fuel at present. The broken lines indicate the results of a similar calculation with 20% enriched fuel containing 164 g U-235 per element. This was the amount deemed equivalent to 150 g of 93% enriched uranium. The "shooting-in" calculations with 160 g and 170 g, respectively, of 20% enriched uranium per element are shown in the figure as well.

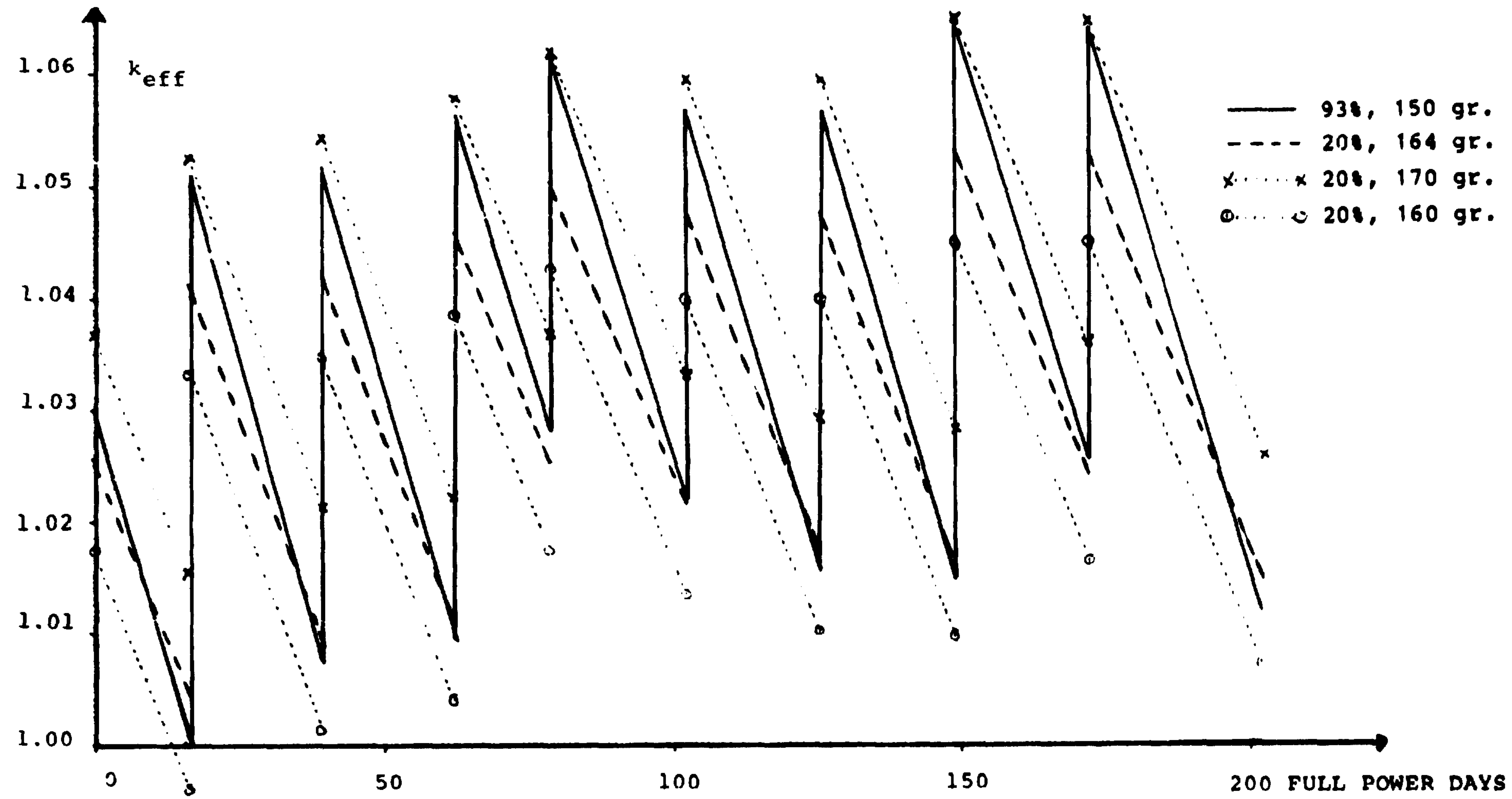


Fig. 4. k_{eff} versus time for various fuels.

REFERENCES

HØJERUP, C.F. (1976), The Cluster Burnup Programme CC^ and a Comparison of its results with NPU-experiments. Risø-M-1898.

LINDSTRØM JENSEN, K.E. (1970), Development and Verification of Nuclear Calculation Methods for Light-Water Reactors. Risø-Report No. 235.

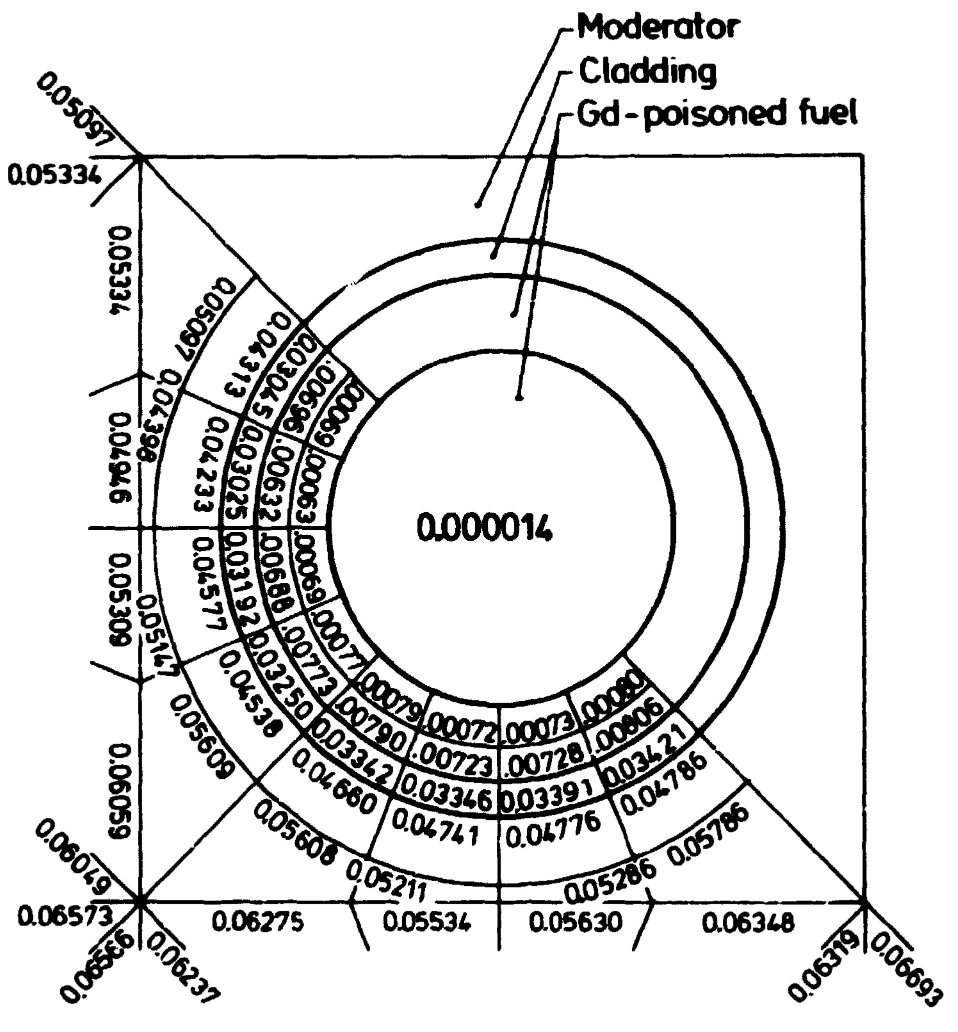
2.6. Response-Matrix Calculations on Fuel Boxes

The multigroup response matrix program REPRO-FLUSO has been developed in the first instance to obtain a detailed flux picture based on transport theory in a fuel box containing different types of pin cells. The obvious extension to other components such as water gaps and control blades has yet to be developed.

The variables coupled by the response matrices are the half space interface currents and variables representing net currents along the interface. These variables operate at a number of Gaussian points along the cell sides. The internal flux response is calculated by collision probabilities in 80-150 of the subregions.

The program has been tested on a BWR Lattice cell benchmark problem from the Nuclear Energy Agency Committee of Reactor Physics, NEACRP: a 9-pin supercell with a control burnable poison pin prepared for 6 energy group calculation. The results compare favourably with the solutions submitted by various countries and in particular with the "best estimate" solution obtained by Winfrith through a thorough analysis (Halsall, 1976).

The response matrices used in this problem were then applied to a 16-pin supercell with two diagonally adjacent poison pins. This configuration was introduced in a NEACRP-benchmark (Maeder and Wydler, 1982) on burnup calculation for a BWR Lattice containing adjacent poisoned fuel rods. Although several solutions were submitted to the committee none of these could throw direct light on the mutual shadowing effect. In spite of the fact that geometry and cross sections are derived from a different problem the REPRO-FLUSO solution could give a relevant snapshot of the begin-of-life situation with regard to the mutual shadow effect of the two poison pins. In Fig. 5 this effect is illustrated by presenting the flux distribution in the strongest absorbing energy group, no. 6, in and around one of the poison pins. The direction to the symmetrically arranged second poison pin is upward and to the left. The flux tilt in this direction is 14 per cent across the poison pin. On this tilt an oscil-



<u>Fig. 5</u>. Thermal flux in and around a Gd-poisoned fuel pin.

lation is superimposed round the perifery caused by the moderator being mainly concentrated in the cell corners. From the two cases solved, it can be deduced that the total absorption in all groups considered in two adjacent poison pins is the double of the absorption in a single pin reduced by 5-6 per cent depending on the kind of normalisation used.

It may be concluded that the mutual shadowing effect in the unvoided case considered is moderate because of the diagonal arrangement, where an efficient separation is established by the moderator in the corners of the four adjacent cells.

REFERENCES

HALSALL, M.J. (1976). Review of International Solutions to NEACRP Benchmark BWR Lattice Cell Problems. AEEW-R 1052.

MAEDER, C. and WYDLER, P. (1982). Burnup Calculation for a BWR Lattice with Adjacent Poisoned Fuel Rods. NEACRP-A-521.

2.7. Three-dimensional PWR Calculations

The computer program for PWR core calculations with combined neutronics and thermal-hydraulics is called ANTI. The primary purpose of this program is transient calculations for safety studies, but it is also used for steady-state and burnup calculations.

Testing of the transient calculations by comparison with measured data has so far been impossible, because no measurements of transient conditions in the reactor core are available to us. Therefore, the power reactor calculations have been concentrated on the steady-state and burnup parts of the program. The reactor chosen for these calcualtions is Ringhals 3, owned by the Swedish utility Statens Vattenfallsverk.

Ringhals 3 is a Westinghouse three-loop plant very similar in design to the plant described in the Reference Safety Analysis

Report RESAR-31. It has a nominal thermal power of 2775 MW and went into operation in 1980. The reactor seems well suited for computer program qualification; it is a typical modern PWR with good in-core instrumentation, and the initial core contains fuel of three different enrichments with burnable poison in the form of boron rods. The owner of Ringhals 3 has supplied both the information needed for the calculations and the measurements from the reactor operation for comparison with the calculation results.

Figures 6 and 7 show the calculated power shapes for different reactor conditions. Figure 7 indicates the normalized fuel assembly power distributions along two different radii of the core. In the 3% power case one control rod bank (the D-bank) is fully inserted with the C-bank at the top of the core, and both of the full power cases have the D-bank at the top of the core.

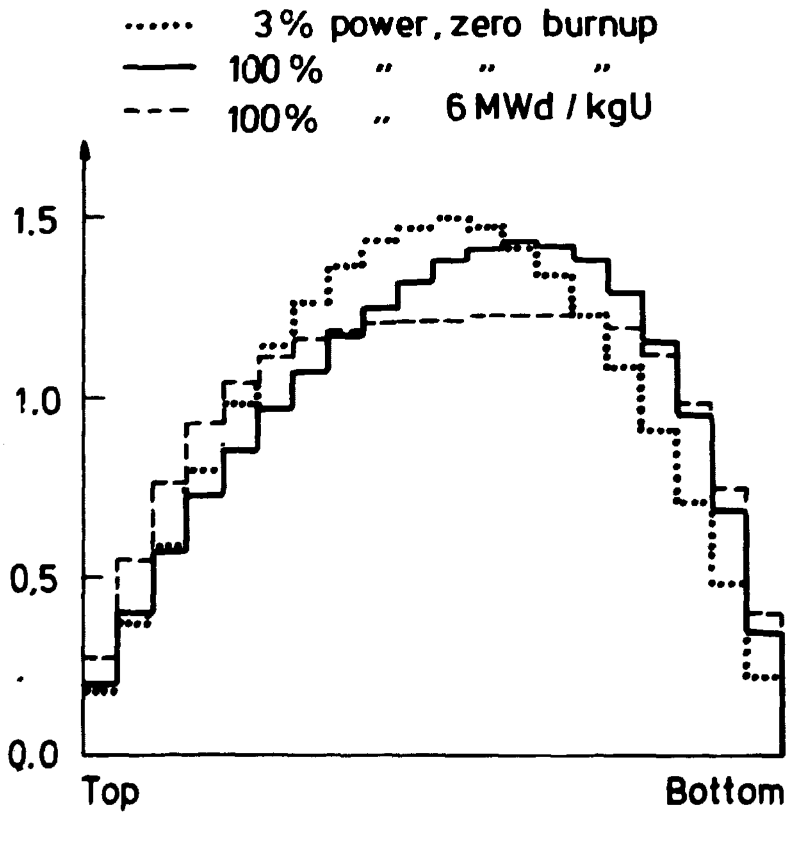


Fig. 6. Axial power distributions for Ringhals 3, calculated by the ANTI program.

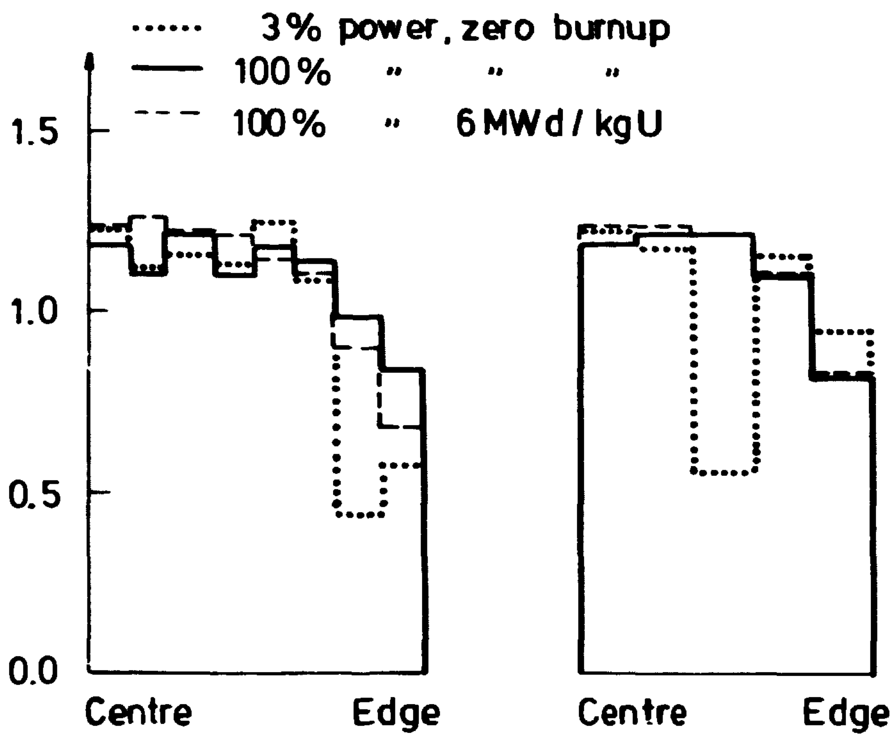


Fig. 7. Horizontal power distributions along two radii for Ringhals 3, calculated by the ANTI program.

From Fig. 6 it is seen that at full power the axial shape is shifted towards the core bottom compared to the shape at low power. Also the flattening with burnup is illustrated. Figure 7 shows the dips in power for the fuel assemblies where the control rods are inserted in the 3% power calculation. The more even power distribution for adjacent assemblies after burnup is caused by the disappearance of the burnable poison.

2.8. Resonance Calculations

The problem of resonance absorption is a central one in the study of the transport of neutrons in a reactor, especially when the geometry of the reactor is complicated. The resonance integral method consists of 2 parts: First, an evaluation of a table of resonance integrals against the potential cross-section for hydrogen in a homogeneous medium. Second, a calculation of the effective potential cross-section for the geometry under consideration. The difficult part of this second step is the calculation of the Dancoff factor, especially when analysing within-pin effects.

A new method which can cope with these difficulties has been analysed, the so-called subgroup method (Roth, 1974). It consists of finding the Lebesgue measure of some representative single resonances, the width of which coresponds to a subgroup, for each isotope. This is done by applying a least-square fit, demanding positive weights, to the above-mentioned table of resonance integrals versus potential cross-section of hydrogen for the homogeneous case. The approximation is done separately for each energy group in the resonance region. Typical 6-8 subgroups are needed to get a good leastsquare approximation for each group.

Then, a collision probability calculation is made for the actual geometry in order to determine the selfshielding factors for each single resonance or subgroup. Finally, the resonance group cross-section is determined by weighting each subgroup resonance with the self-shielding factor and the Lebesgue measure for each energy group.

The advantage of the method lies in the fact that the collision probability can be calculated even in complicated geometries. Furthermore, the collision probability calculation, the most expensive part, needs only to be made for each subgroup.

Until now the method has been tested only against pin cell geometry and compared with calculations applying Dancoff-formalism. The comparison has turned out quite well but it will be more thoroughly tested for within-pin effects and possibly the method will be inplemental in the standard cross section processing codes at Risø.

REFERENCE

ROTH, M.J. (1974), Resonance Absorbtion in Complicated Geometries, AEEW-R921.

2.9. New Method for Three-dimensional Calculations

A new 3-dimensional nodal programme for power calculations is under development. The programme is based on the Nodal Expansion Method (NEM), which uses the average flux in each node and the partial currents at each boundary as primary variables. The partial currents are calculated by means of additional 1-dimensional diffusion calculations within each node, where the 1-dimensional fluxes are expanded after polynomials to various order (2-4). Furthermore, the transverse leakages are expanded to various orders (0-2). A very strong coupling between 1-dimensional and 3-dimensional calculations is achieved, and this gives good convergence properties for the method.

The method has been programmed in a code called NEM with the possibility of using various flux expansion order, transverse leakage expansion order, iteration techniques etc. The programme has been tested on various benchmark problems with good results. The convergence properties has especially been tested on the 2-dimensional IAEA-benchmark (Fig. 8) and some of the results are shown in Fig. 9. The conclusion is that we have to use fourth-order local flux expansion and first- or second-order transverse leakage expansion order to achieve satisfactory results for large nodes.

Unfortunately, calcuations with high-order expansions are rather expensive, so a version of the programme, which uses low-order expansions in the beginning and then changes to high-order expansions has been programmed. With this procedure a gain in computing time by a factor of 2-3 has been achieved. The method has been tested on several benchmarks both in 2 and 3 dimensions and with a node size equal to the size of a fuel element (~ 20 cm). An accuracy better than 0.1 per cent in the eigenvalue calculation and 3 per cent in the power distribution is achieved. Until now only calculations on benchmark problems have been made, but calculations on a real reactor and comparisons with measurements are under preparation.

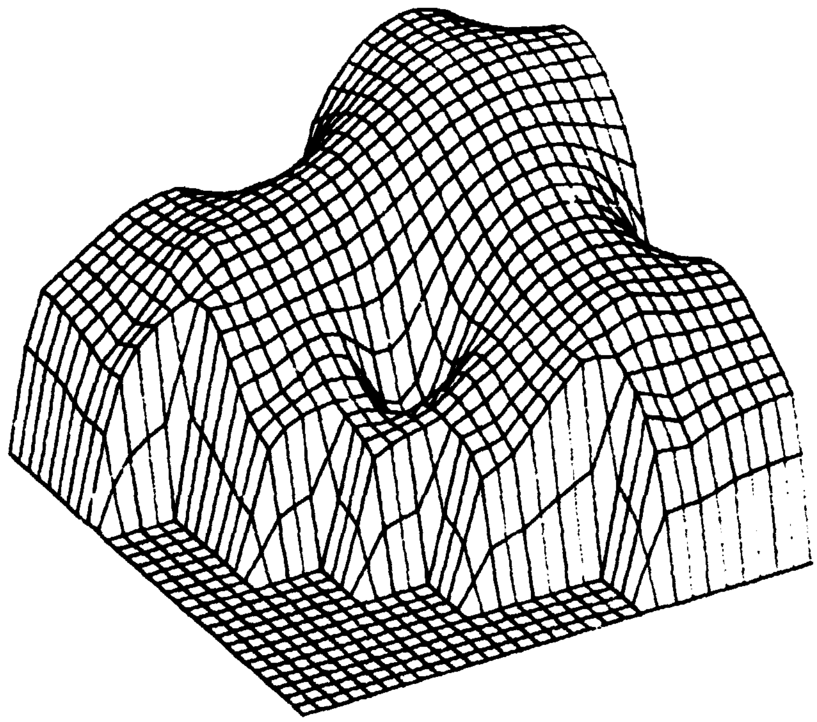


Fig. 8. The radial power distribution for the 2-dimensional IAEA benchmark for one-quarter of the core.

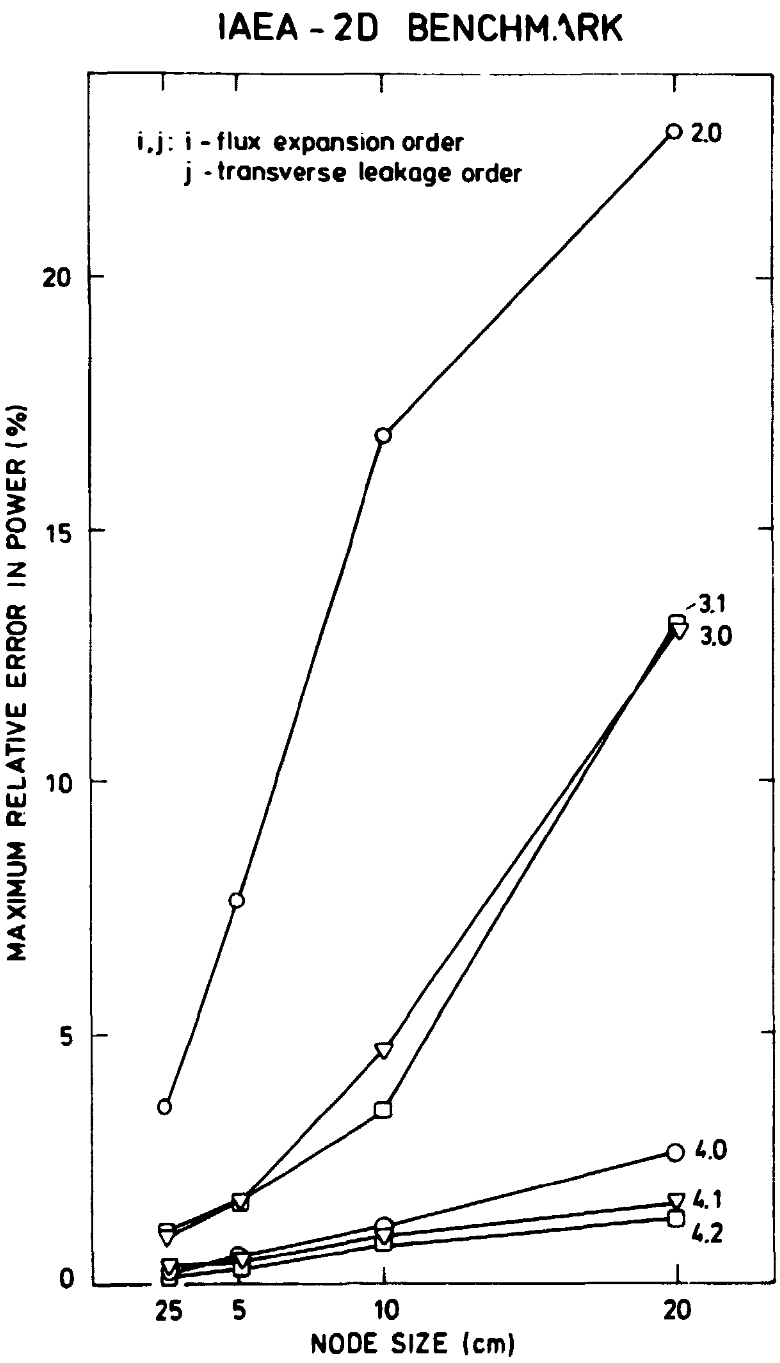


Fig. 9. The maximum relative error in the power distribution as a function of the node size for various expansion orders.

2.10. Severe Accident Analysis

Under a contract with the Danish power utilities, ELSAM and ELKRAFT, the Department has initiated a work on "Severe Accident Analysis" for nuclear light-water reactors.

Two computer codes have been used in the analysis: : MARCH (core meltdown, thermohydraulic response of vessel and containment, molten core-concrete interaction), and HAARM-S (radioactive aerosol transport and retention).

So far, two accident sequences have been analyzed for a Swedish type boiling-water reactor:

a. The T_IW sequence, i.e. total loss of external cooling but with functioning emergency core cooling from the pressure suppression pool. This results in a containment break due to overpressure leading to uncoverage and melting down of the core, which subsequently melts through the reactor vessel, falls to the floor of the containment (drywell) and starts to melt through it.

b. The T_B sequence, i.e. total loss of electric power. This makes the core boil dry and melt down through the vessel to the floor of the containment (drywell), where it starts to melt the concrete floor. After a long period of time the concrete melting builds up an overpressure which makes the containment break.

For the T_IW sequence the analysis indicated that 85% of the aerosol-carried fission products released during core melting will be retained in the vessel, containment, and reactor building. For the T_B sequence the indication is that less than 0.01% of the aerosol-carried fission products will be released from the reactor building as calculations have shown that the containment will remain unbroken for several hours after core meltdown.

Based on a recent General Electric paper, the analysis work has also shown the possibility that for both sequences one should be able to get the release of aerosol-carried fission products down to less than 0.01%. The method is to force the gas carrying the aerosols through the pressure suppression pool where the aerosols are "scrubbed" out of the gas before entering the reactor building.

2.11. The Advanced BWR Emergency Core Cooling Program NORCOOL-II.

NORCOOL-II is an advanced thermohydraulic computer code with input specified geometry for simulation of a loss-of-coolant accident in a boiling water reactor. Until medio 1980 the code was developed as a joint Nordic work within the NORHAV cooperation, but the development after that has been purely Danish.

The basic hydrodynamics were tested successfully using a hypothetical "cold" refill of a 4-m downcomer at a nominal water velocity of 16 cm/s.

Some transient natural circulation experiments have been simulated with surprisingly good agreement (practically no code adjustments). For one of these experiments the NORCOOL-II results are shown together with the measured values in Fig. 11. The discrepancies could be traced back to insufficient heat transfer models.

In order to perform simulations of real "hot" quenching experiments (by reflooding) some minor numerical improvements were made to the wall and fuel-rod heat conduction model. Although this model was made to function satisfactorily it soon became clear that with the present solution method for the hydraulics, which greatly limits the time step size, NORCOOL-II cannot in practice simulate quenchings due to excessive use of computer time. Therefore, an alternative "global" solution method for the hydraulics has been developed which should allow considerably longer time steps and, hopefully, also a shorter computer time per time step. The new solution method is expected to be ready in the beginning of 1983.

2.12. Smal Break LOCA Analysis, SÄK-3

The SÄK-3 project is an internordic reactor safety project carried out under the auspices of the Nordic Liaison Committee for Atomic Energy (Nordic abbreviation: NKA). The objective of the project is to provide one or more computer codes suitable for small break loss-of-coolant accident analysis. Four different codes have been studied during 1982. The American code RELAP5 has been studied by Finland and Sweden, the TRAC-PF1 code, also American, has been studied by Denmark and Norway, the Norwegian code RAMONA-II-PWR by Norway, and the Finnish code SMABRE by Finland.

Until the implementation of TRAC-PF1 on Risø's Burroughs 7800 was completed ultimo September the TRAC calculations made by Risø were carried out on the CDC Cyber 172 installation in Studsvik, Sweden. Two major test cases, the LOFT experiments L3-6 and L3-5, have been calculated, and the results have been compared to the experimental data and to the RELAP5 calculations of Finland and Sweden. The comparison to data showed that especially the choked flow model but also the heat transfer model for reflux-condenser mode seem questionable and require a deeper study. Despite that, the calculated trends follow the measurements rather well.

The TRAC code is not free from errors, but its operation is stable, it is relatively easy to use, and has so far produced fair results.

2.13. Heat Transfer Correlations, SÄK-5.

Like SÄK-3, the SÄK-5 project is an Internordic reactor safety project under the auspices of the NKA. The objective of the SÄK-5 project is to provide a set of reliable heat transfer correlations for use in advanced best estimate computer programmes for LOCA analysis. The effort at Risø has been put in the investigation of the so-called transition boiling regime in which the heat flux from wall to coolant decreases with increasing wall temperature. Despite the highly unstable characteristic of this regime in a heat flux controlled system like a nuclear reactor it is of importance during the strong transient phases of a loss-of-coolant accident and the emergency cooling phase that follows.

Some effort has been given to a study of the flexibility of the TRAC-PF1 code with respect to change heat transfer correlations and the use of the code to test heat transfer correlation.

The change of heat transfer correlations for the wall to coolant seems to be fairly easy, but not for the interfacial heat transfer, which controls the mass transfer and thereby the void fraction, which again has a feedback effect on the wall heat transfer correlation. It is important to bear this in mind. It is the total set of built-in correlations that determines the local parameters used in the tested correlation. Therefore experiments should be selected that cover only one heat transfer regime in order to minimize the influence of correlations not in question. This has been confirmed by some calculations with TRAC upon some transient experiments. These calculations indicate also that selected experiments should be of the steady-state type.

Development of small independent codes to test some correlations against experiments may be necessary. This is particularly the case with correlations in the transition boiling regime. Reliable test results may be obtained only from experiments that have temperature-controlled surfaces. Such experiments have been found only recently.

2.14. Emergency Core Cooling Experiments

A test facility has been erected with the purpose of investigating the parallel channel interaction during flooding. The test loop consists of 3 parallel tubes connected through a lower and an upper plenum. The tubes are directly heated. Each is equipped with the following instrumentation:

1 flowmeter, 20 thermocouples, 3-9 pressure transmitters.

2 different types of experiments have been made:

1. Quenching (flooding), 1-3 parallel tubes
   Initial temperature: ~ 600^{o}C
   Inlet velocity: 0.00 - 0.15 m/s
   Inlet temperature: 20 - 80^{o}C
   System pressure: 1 bar
   Power: 0 - 3 kW

2. Natural circulation

A few stationary natural circulation experiments have been performed. However, the main interest was to investigate the transient natural circulation. The loop is filled with water at constant initial temperature. At time > 0 one of the tubes is heated with constant power (a step function) and the circulation starts (Fig. 10).

The experiments have been compared with predictions from the computer code NORCOOL II (Fig. 11).

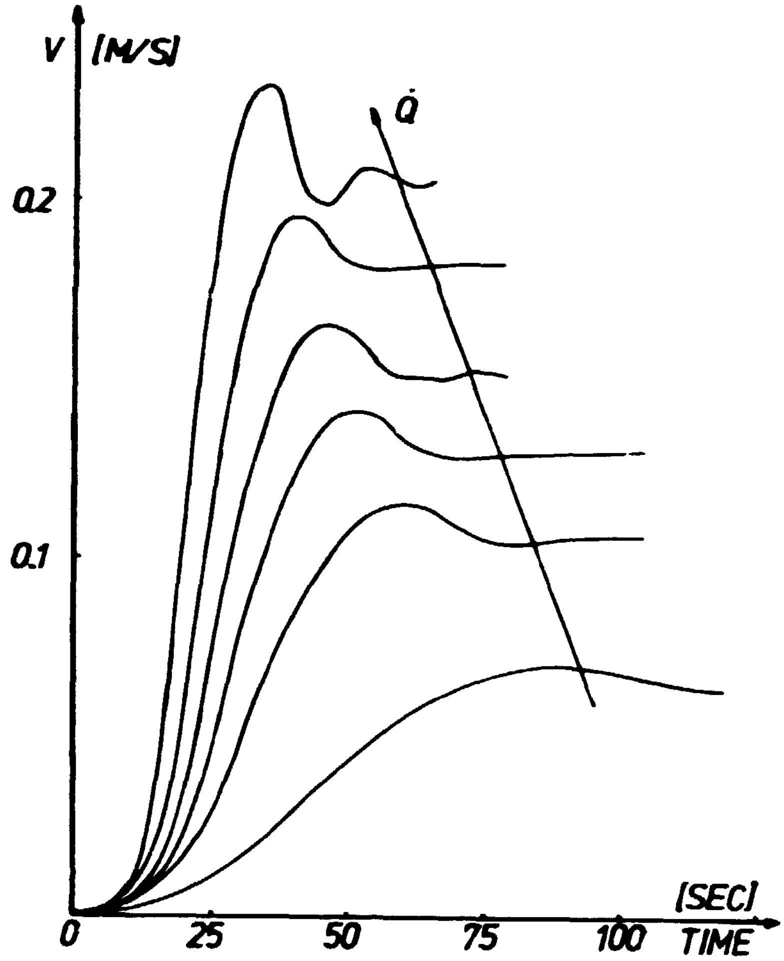


Fig. 10. Transient natural circulation. Inlet velocity V versus time with different power Q. Q = 0.80, 1.95, 2.70, 3.65, 5.5, 7.43 [kW]. The loop consists of two (ϕ15x1) vertical and parallel channels of length L= 4.709 [m]. Initial temperature 20 [^{o}C].

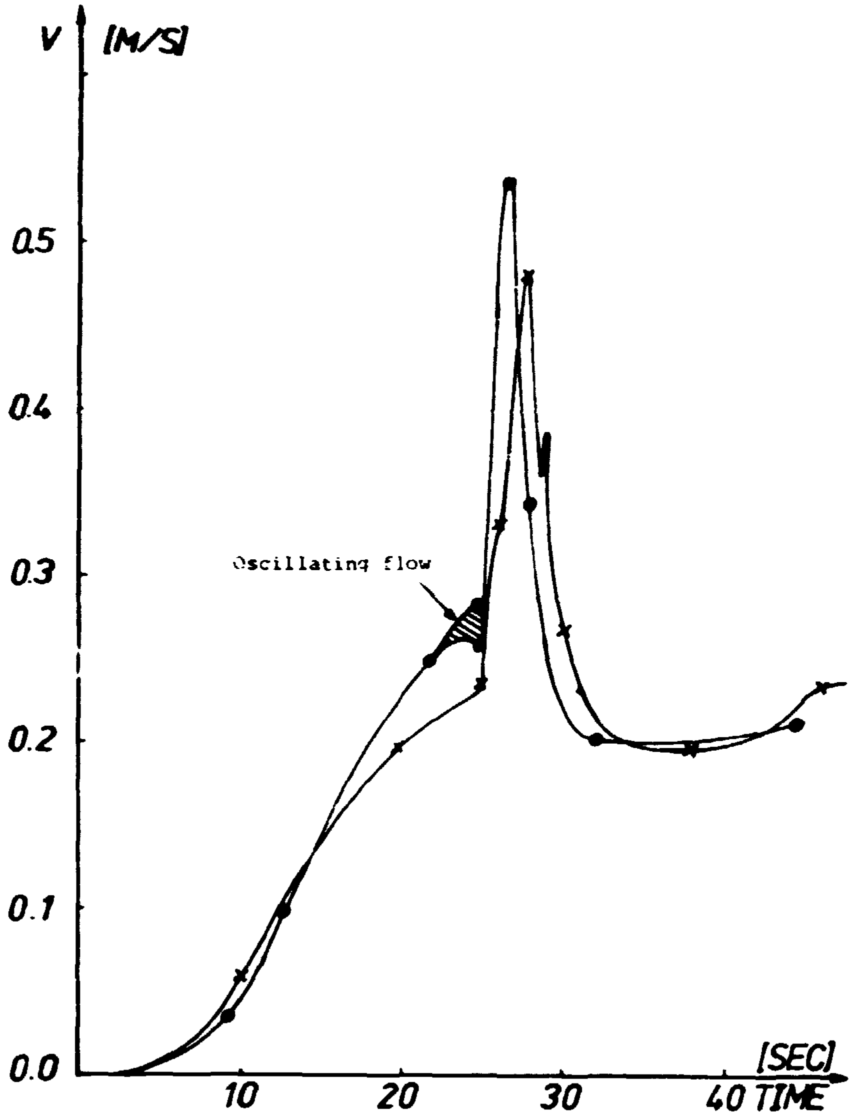


Fig. 11. Transient natural circulation. Velocity versus time. Comparison between experiment and calculation. Initial temperature 20 [°C]. Power 10.0 [kW].
o experiment x calculation

2.15. Experimental Study of Rewetting and Quench Phenomena

An experimental programme has been started with the purpose of investigating the behaviour of electrically heated fuel pin simulators during reflooding with special emphasis given to the pin composition. The experiment includes directly heated pins, conventionally indirectly heated pins, and advanced indirectly heated pins containing uranium dioxide as filler material.

A series of reference measurements with a directly heated tubular test section has been made. The fuel pin simulators are tested in an annular geometry with only one pin at a time. The experiments with the first type of pins, the directly heated pin, were finished late in 1982.

The test parameters have been:

Initial pin temperature: 600, 700, and 800°C
Coolant inlet temperature: 20 - 95°C
Coolant inlet velocity: 1 - 10 cm/s

The two remaining types of pins will be tested in 1983. The experimental results will be compared with predictions from calculations with the NORCOOL II computer code.

The programme is a part of the Commission of the European Communities "Indirect Action Programme on the Safety of Thermal Water Reactors" and is partially financed by the CEC.

2.16. Blowdown from High Pressure Systems

In several industrial applications liquefied gases, such as clorine, ammonia, butane and propane, or natural gas/oil mixtures, are kept under high pressure. To be able to estimate the consequences to the environment, in the event of a pipe or tank leak (blowdown), it is necessary to have a reasonably accurate model which can predict the release velocity, the duration of the leak and the amount of liquid in proportion to gas which enters the environment. The liquid will be released to the atmosphere as small drops, and because the spreading of liquid in the atmosphere depends on the drop sizes produced by the blowdown, the model must also be able to predict the size of these drops. The work will also include steam/water mixtures.

In September a Ph.D. project was begun in which the blowdown situation was investigated experimentally and theoretically. Special efforts will be made to measure the sizes of the drops produced by the blowdown, and to develop a model which can predict these drop sizes.

2.17. Heat Exchangers

A research on industrial heat exchangers was started in September 1982 in cooperation with a Danish boiler manufacturer. A survey was made of which investigations were needed. It was discovered that the problems about dirt resistance to heat transfer took first precedence, and a fouling test rig has been designed to be placed in operation in the near future.

Also the possibilities of buying commercial heat exchanger design programs are under investigation, but these programs seem to be not as valuable when the heat exchangers are designed according to other than American standards. For that reason the development of a heat exchanger design program has been started.

2.18. The Temperature Calibration Laboratory

The Temperature Calibration Laboratory was authorized in 1978 by the Danish National Testing Board to carry out certified calibrations of temperature sensors in the -150°C to 1100°C range according to the International Practical Temperature Scale IPTS-68. The standard thermometers in the Laboratory are traceable to the National Physical Laboratory, England.

During 1982 the Laboratory had performed 106 jobs for external customers and 9 for other Risø departments. In all, 340 thermometers ranging from liquid-in-glass models to advanced digital types and 14 thermostats had been calibrated during the year. The calibrations had been made in the temperature region from -90°C to 1100°C which covers the main part of the range authorized.

2.19. Preliminary Investigations of Pressurized Fluidized-bed Combustion

A preliminary investigation was started in order to gain general theoretical and experimental knowledge of the state of the art of fluidized-bed combustion of coal in combined cycle plants. The project is under contract with the Ministry of Energy. The main efforts are planned to be centred on pressurized and circulating fluidized beds.

A report is in preparation describing the fluidized-bed systems in general, pressurized fluidized-bed (PFB) in existence or being constructed, sulphur retention by addition of dolomite or limestone to the bed, material erosion and corrosion in gasturbines, status of fly ash handling and use for different purposes, and finally costs of plants and operating costs.

During 1982 an experimental fluidized-bed working at atomospheric pressure (AFB) has been constructed. The AFB test rig is shown on Fig. 12. The bed diameter is 0.5 m, fluidization velocity is 1-2 m/s, bed temperature 800-900^{o}C, thermal power about 200 kW_t, and consumption of coal 25-30 kg/h depending on the quality of the coal.

This small-sized AFB is established as a test facility for development and testing of instruments and technical details important for AFB and PFB.

An experimental programme is planned for the first half year of 1983. Emission of NO_x and particles will be measured. The retention of sulphur by adding limestone or dolomite to the bed will be investigated using coal of differing sulphur content.

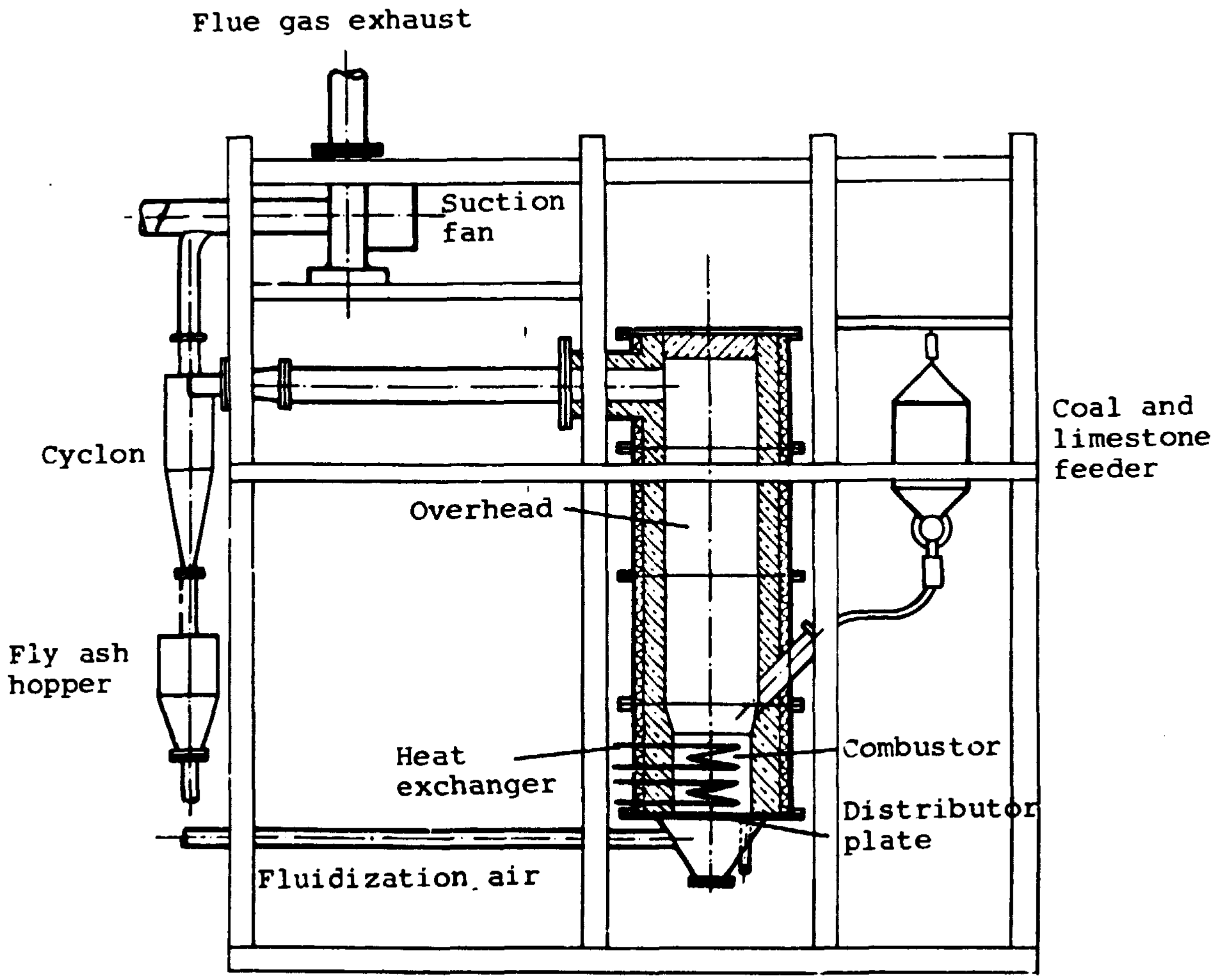


Fig. 12. Atmospheric fluidized-bed test rig.

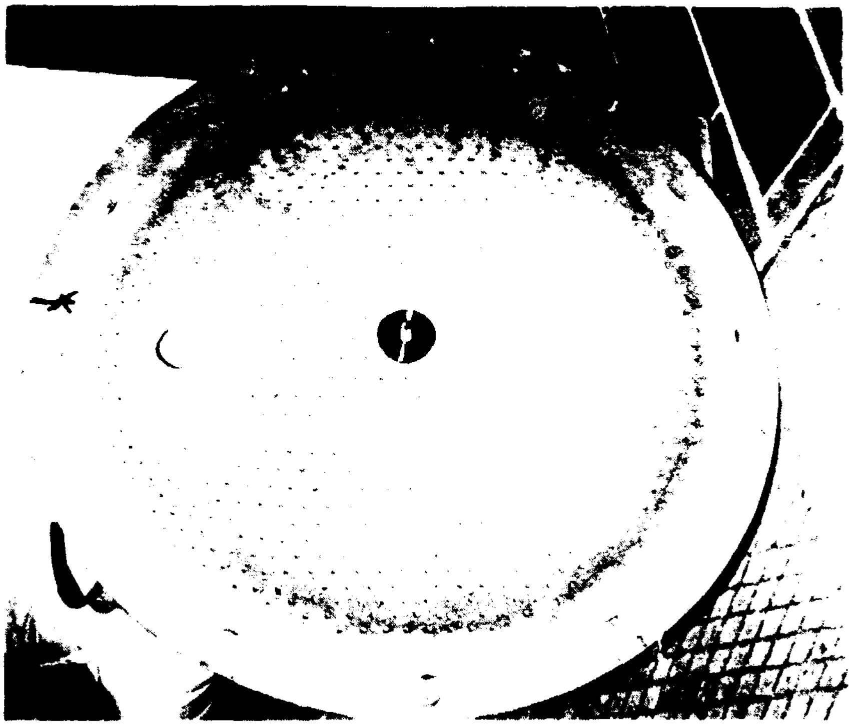


Fig. 13. Perforated distributor plate for atmospheric fluidized-bed test rig.

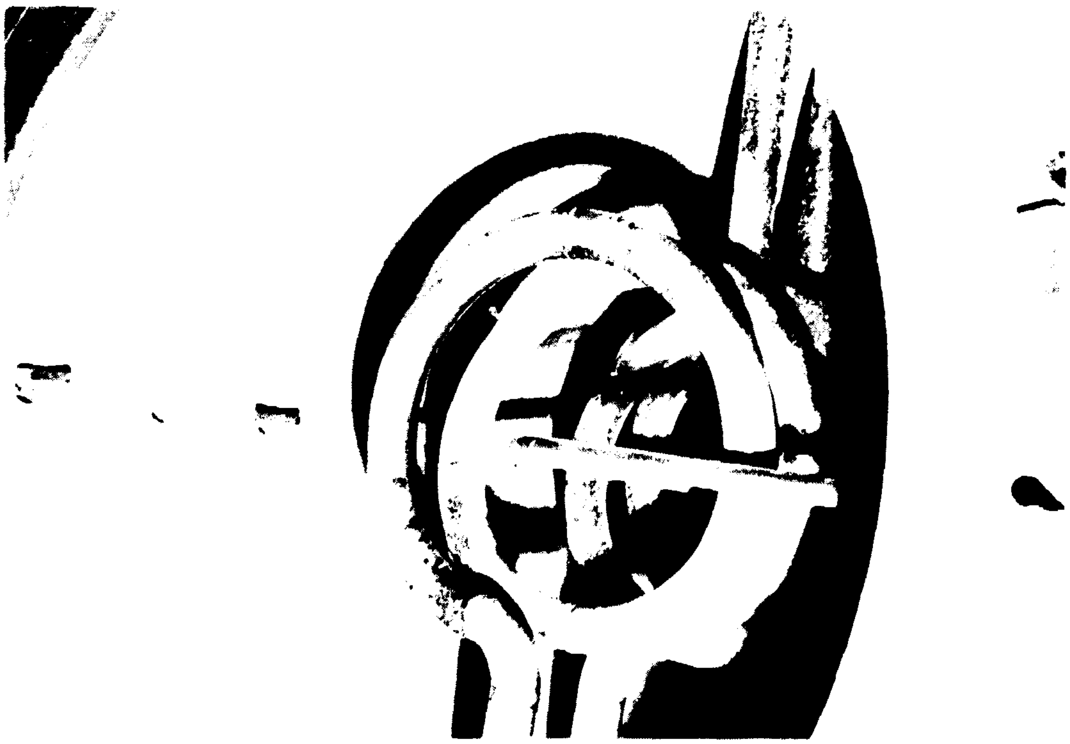


Fig. 14. Heat exchanger tubes in bed, capacity 100 kW.

2.20. Oil and Gas Reservoir Models

The project on oil and gas reservoir modelling has expanded in 1982. The main reason for this was the recognized need for reservoir simulation in connection with the development of the hydrocarbon reservoirs in the Danish part of the North Sea. As a consequence a reservoir group as a separate entity within the Department was established during 1982.

Work has been going on in a number of separate but related fields, and partly in fulfilment of contracts awarded by the Danish Ministry of Energy.

Black-oil simulators have been studied and a 2-dimensional black-oil simulator, obtained from Comtech Inc. USA, has been modified, tested, and documented. An improved version of the 3-dimensional black-oil simulator, BETA 3A, developed by Institutt for Energiteknikk (Norway), has been implemented and tested, and will be used for further work in the black-oil field. The work has been carried out jointly by Risø and The Laboratory of Energetics at the Technical University of Denmark in collaboration with other laboratories at the University and with The Geological Survey of Denmark, The Danish Energy Agency, and Danish Oil and Gas Ltd.

A study has been undertaken in order to identify the needs for more advanced simulators, mainly with enhanced recovery schemes in mind. In this connection it has been decided to undertake a preliminary study to define the contents and structure of a compositional simulator for these purposes. This work has been undertaken in collaboration with the partners mentioned above.

Further studies concerning the problems around the representation of fracture permeability in simulators have been carried out, mainly at the Technical University.

Field studies for a couple of hydrocarbon reservoirs in the Danish part of the North Sea have been carried out in collaboration with The Danish Energy Agency and The Geological Survey of Denmark.

2.21 Energy Storage

The Danish aquifer storage project is a joint project of The Laboratory for Energetics of The Technical University of Denmark, The Geological Survey of Denmark, and Risø National Laboratory, financed by the Danish Ministry of Energy.

During 1982 the project has been concentrating on getting the test facility north of Copenhagen ready for use.

Late in the autumn a number of hot water injections were performed and then had to be suspended until the coming spring due to lack of surplus heat.

The Department's participation in the project was formerly mainly limited to the development and use of reservoir simulators. During 1982, however, the main effort has been directed towards the development of software for control and data-recording purposes.

2.22. Focusing Solar Collector

The feasibility study concerning solar heating of buildings by means of large focusing collectors with weather shields has been terminated. The final report is now available (Thomsen, 1982).

The study was financed in part under a contract awarded by the Danish Ministry of Energy.

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2.23. Digitizing of Neutron Radiographs

In order to extract the maximum possible information from the neutron radiographs, some radiographs have been digitized at a military research institute.

With the equipment it was possible to digitize pictures into 512 x 512 pixels (picture elements) each containing 8 bits. Using these pictures a student from Aalborg University Centre has completed the requirements for a master's thesis "Image Processing of Neutron Radiographs". In the thesis different forms of filtering enchancement and restoration were investigated.

With the equipment it was possible to present the pictures with pseudo-colours.

Equipment to digitize pictures into at least 512 x 512 pixels has been ordered, and is expected to be installed in one of Risø's PDP-11 computers.

Programmes for enchancement and restoration of the radiographs are under development. In addition programmes for quantitative measurements (densities, dimensions, areas, particle counting) are also being developed.

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APPENDIX A

Computer Programs

Description	Code name
Reliability Analysis	
System Reliability	MOCARE

Calculates reliabilty characteristics for systems of any kind using Monte Carlo simulation with or without variance reduction. Very flexible modelling is obtained due to the application of "sub-system models". This makes the program particularly well suited to the analysis of systems with complex design and/or operation. Likewise, it makes it possible to design approximation-free models of multistage failures and phased missions. User-friendly interactive input code is available, and a graphical display of both component and system performance can be obtained. Computer time may be excessive in special cases, for instance, in the case of calculation of the probability density function for the unavailability of a system, if the magnitude of the failure rates varies by several decades.

Component Reliability	ANPEP

Calculates the probability of failure of a structure for stress-strength models used in probabilistic fracture mechanics. The parameters describing stress and strength are given by probability density functions. The calculations are based upon numerical integration in one or several dimensions. In the present version of the program the number of parameters is limited to six.

Reactor Physics and Dynamics

Group Cross Section Generation — SIGMA

SIGMA generates a 76-group neutron cross section master tape for reactor physics calculations from UKNDL. Resonance absorption and thermal scattering cross sections are treated separately in the program RESAB and NELKINSCM.

Resonance Absorption — RESAB

RESAB generates self-shielded, few-group neutron cross sections in the resonance region using collision probability calculations in several thausand groups.

Scattering Data — NELKINSCM

NELKINSCM generates multigroup neutron scattering cross sections in the thermal region. The program uses the NELKIN model for H and D and the Free Gas Model for other nuclides.

Pin and CLUSTER Cell Calculation — CCC

CCC is a 76-group, collision probability, pin cell calculation program. In addition, collision probability cluster cell calculations and burnup calculations in an arbitrary number of groups can be performed. The program has facilities for condensation and homogenisation of cross sections for further use in fuel element calculations.

Fuel Element Calculations for LWR — CDB

CDB performs burnup calculations for LWR fuel elements. Few-group cross sections to be used in overall three-dimensional flux distribution calculations by means of coarse-mesh methods can be generated. The program uses multi-group collision probability pin cell calculations with burnup, and XY-diffusion theory calculation for calculation of power distribution within the element.

Flux Calculations with Burnup DBU

DBU determines its diffusion cross sections by interpolations in precalculated tables. The cross sections are given as functions of up to 3 variables: burnup, power level, and void contents. For a given series of time steps the program calculates flux, power and burnup distributions by means of diffusion theory.

Steady-state Neutron Diffusion Theory

The steady-state neutron diffusion equation for calculation of reactivity and flux-power distribution can be solved in two and three dimensions with different approximations.

TWODIM is a two-dimensional difference equation program, using center mesh points. (X,Y), (R,Z), and (R,θ) geometry can be used. Arbitrary number of groups. TWODIM

TVEDIM is a two-dimensional difference equation program, using corner mesh point. (X,Y), (R,Z), and (R,θ) geometry can be used. Arbitrary number of groups. TVEDIM

FEM uses the finite-element approximation for two-dimensional calculations. Up to 5th order has been used. (X,Y) geometry and arbitrary number of groups. FEM

DC4 is a three-dimensional difference equation program using corner mesh points. (X,Y,Z) geometry and arbitrary number of groups. DC4

FEM3D uses the finite-element approximation for three-dimensional calculations. Up to 3rd order can be used. (X,Y,Z) geometry and arbitrary number of groups. FEM3D

SYNTRON is a three-dimensional single-channel flux synthesis program. (X,Y,Z) geometry and arbitrary number of groups. SYNTRON

<u>Three-dimensional Steady-state BWR Simulator</u> NOTAM

NOTAM is a three-dimensional steady-state BWR simulator. It combines a neutronic module based on FLARE-type nodal theory with a hydraulic module based on multichannel, one-dimensional flow. The program has facilities for burnup calculations.

<u>Three-dimensional PWR Simulator</u> ANTI

ANTI is a three-dimensional PWR simulator. It combines a neutronic module bassed on FLARE-type three-dimensional nodal theory with a hydraulic module based on subchannel flow. The program can perform both steady-state and transient calculations for PWR core. In addition it has facilities for burnup calculations.

<u>Three-dimensioanl BWR Transient Simulator</u> DANAID

DANAID is a three-dimensional BWR core transeint simulator. It combines a neutronic module based on FLARE-type three-dimensional nodal theory and a hydraulic module based on multichannel one-dimensional flow.

<u>Simulation System for Dynamic Processes</u> DYSIM

DYSIM is a standardized program system for simulation of dynamic processes which are described by a mixed set of differential and algebraic equations. The state variable concept is used with the possibility for shifting variables between state and algebraic variables. True delay simulation may be included. Fortran programming by subroutines which are bound to DYSIM. Standardized input with specification of initial conditions, pertubations, control parameters, and output tabulations.

<u>Simulation of PWR Plant Dynamics</u> PWR-PLASIM

Dynamic model of a PWR power plant for transient calculations at "normal conditions". The primary circuit consists of one loop with reactor, pressurizer, steam generator, and pump. One-dimensional calculation of nuclear power coupled to hydraulic equations, and one-dimensional calculation of heat transmission to the secondary side of the steam generator. Secondary side with one-dimensional steam line, turbine model, and feedwater system. Main control and protection systems are included.

<u>Simulation of BWR Plant Dynamics</u> BWR-PLASIM

Dynamic model of a BWR power plant for transient calculations of "normal conditions". One dimensional calculation of nuclear power coupled to hydraulic equations. Forced recirculation in reactor with external pumps. The steam load system consists of a one-dimensional steam line, turbine model and feedwater system. Main control and protection systems are included.

<u>Simulation of PWR Plant Dynamics Including Detailed Turbine Model</u> BWR PLANT

Dynamic model for a BWR power plant for transient calculations. The reactor model consists of a one-dimensional, multigroup, finite-element neutronics model, and a one-dimensional model of hydraulics. The reactor model is coupled to a detailed turbine model consisting of a high-pressure turbine, a moisture separator, a reheater, condenser, feedwater heaters, and extractions for district heating.

Fuel Management SOFIE

SOFIE is a program which can determine economically optimal refuelling strategies. It is a combination of a one-dimensional reactor physics model of a reactor core and a module for calculating fuel economics. Optimization is carried out by means of linear programming.

Reactor Thermo-hydraulics and Accident Analysis

Reactor Steady-state Heat Transfer and Hydraulics SDS

Subchannel core heat transfer and hydraulics. Best-estimate, steady state, flow, and enthalpy, BWR and PWR, non-equilibrium, drift-flux, boundary value solution technique, saturated steam only. Arbitrary number of subchannels.

PWR Blowdown TINA

Calcualtion of low rates, void fractions, and liquid and fuel rod temperatures in the PWR core. Best estimate, subchannel approach, non-equilibrium, drift flux. Arbitrary number of subchannels. Saturated steam only. Fuel rod damage is not considered.

BWR Top Spray Emergency Core Cooling CORECOOL

Best estimate. Single loop geometry, spray cooling, one-dimensional non-equilibrium two-fluid plus falling films, detailed radiation heat transfer. Results: Two-phase hydraulic and thermal state, temperatures of fuel, cladding, and fuel element box.

<u>BWR Emergency Core Cooling</u> NORCOOL-I

Best estimate. BWR-vessel geometry with single fuel element. Spray injection and normal bottom reflooding. One-dimensional numerics, saturated steam drift-flux in continous water, non-equilibrium two-fluid plus falling films in continuous steam regions, explicit two-phase level and quench front tracking. Detailed radiation heat transfer. Results: Two-phase hydraulic and thermal state, temperatures of fuel, cladding, and fuel element box. (Does not calculate blowdown).

<u>PWR Blowdown and Emergency Core Cooling</u> TRAC-PF1

Best estimate system code. Any system geometry can be modelled with pipes, tees, valves, pumps, etc. One-dimensional numerics, three-dimensional numerics optional for reactor vessel component. Non-equilibrium two-fluid hydraulics. Moving mesh in fuel rod heat conduction calculation gives quench front tracking. Point kinetics. Results: Two-phase hydraulic and thermal state, temperature of fuel, cladding, and heat structures. (Program developed by Los Alamos, USA).

<u>PS-Containment LOCA Response</u> CONTAC-III

Calculates pressure- and temperature response to LOCA for a pressure suppression-type containment. The code includes models of the reactor vesel and two containment volumes. The code assumes thermal equilibrium and is based on a quasi-stationary calculation.

<u>Containment Response to Core Meltdown</u> MARCH

Models a variety of physical processes consistent with the phenomena expected to be associated with meltdown accidents in light-water power reactors: heat-up and boil-off of water in reactor vessel, clad oxydation and slumping of the fuel, vessel meltthrough, interaction of core debris with water and concrete and hydrogen burning. Results: Containing building response: temperature, pressure and intercompartment flows, and release of fission products from fuel. (Program developed by Battelle Columbus, USA).

Fission Product Transport and Removal — CORRAL

Fission product removal processes in the containment as leakage, deposition and spray are modelled. The containment thermal hydraulics and fission product source terms required for the program are provided by MARCH. Results: Quantity of airborn fission products available for release to the environment at any time. (Program developed by Battelle Columbus, USA).

Aquifers and Reservoirs

Heat Storage in Aquifers - Linear Finite-Element Model — PORFLOW

x-y or r-z geometry. Calculates the transient pressure and temperature field during heat storage/withdrawal. Simplified representation of the temperature field by a hot and cold zone.

Heat Storage in Aquifers - Two-dimensional, Quadratic Isoparametric Finite Elements — D2AQ

Detailed temperature and pressure field during injection/withdrawal in a porous medium. z-y and r-z geometry. (Program from Laboratory for Energetics, Technical University of Denmark.)

Ground Water Pollution Dispersion — SWIP

Three-dimensional, finite-different model for transient calculation of pressure- and temperature distribution, and transfer of fluid and soluble matter through a porous matrix. x-y-z and r-z geometry. (Program from U.S. Geological Survey).

Petroleum Reservoir Simulation DM

Black-oil reservoir simulator. Two space dimensions (x,y), three components and three phases including water. Calculates oil in place, pressure, flow rate and saturation for each phase, production rates and recovery factors. (Code developed by Comtech Inc., USA).

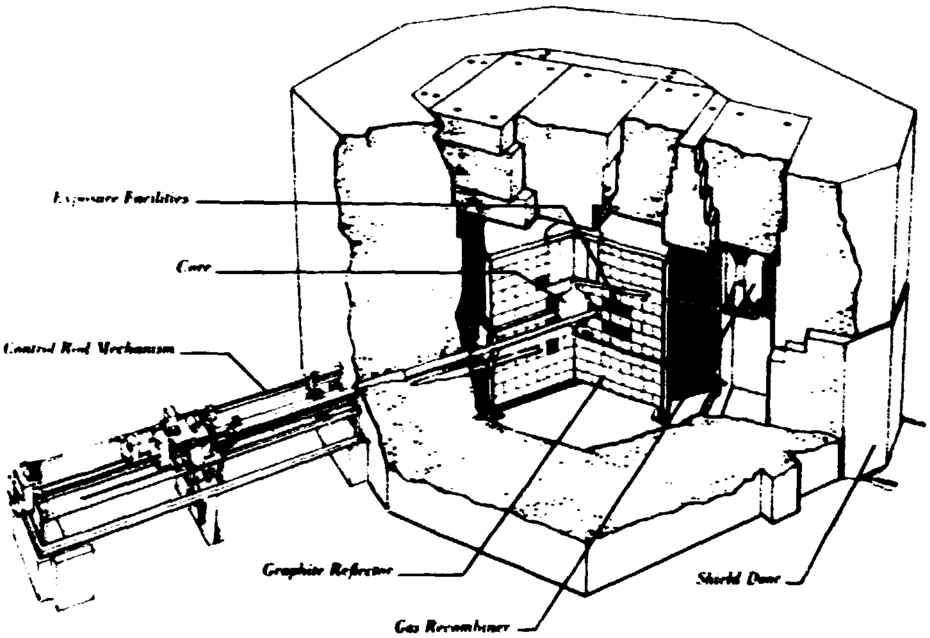
Petroleum Reservoir Simulation BETA 3A

Black-oil reservoir simulator. Three space dimensions (x,y,z) or R,Z-option. Three components and three phases including water. The code calculates oil in place, pressure, flow rate, and saturation for each phase, production rates and recovery factors. (Code developed by IFE, Norway).

APPENDIX B

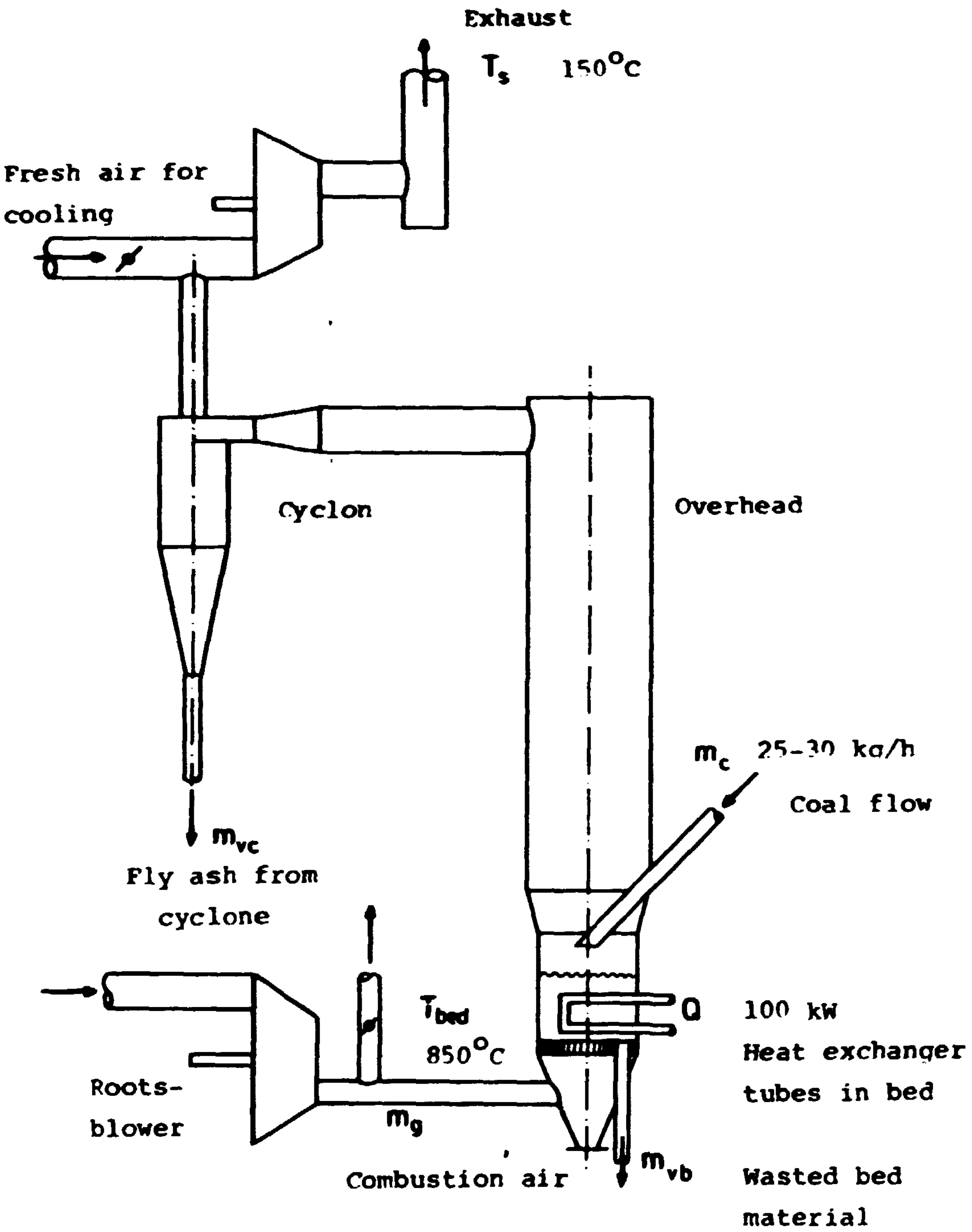
Test Facilities

Danish Reactor No. 1



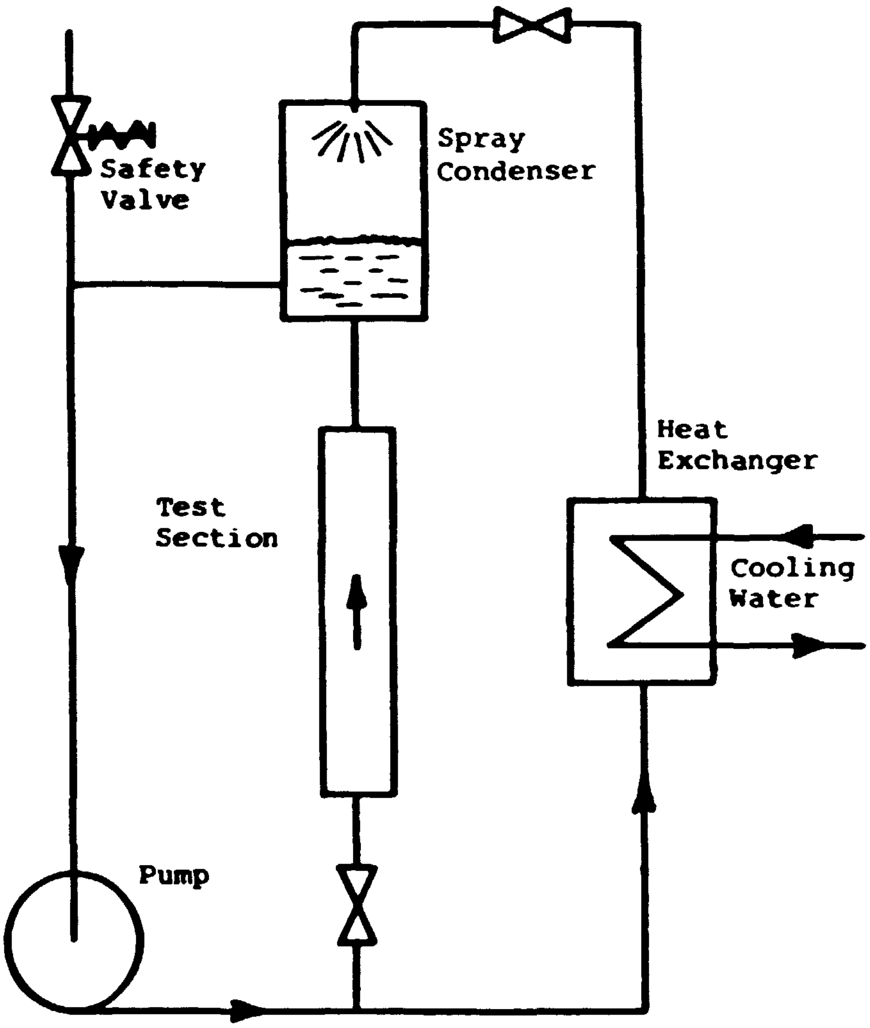
Type	Aqueous-homogeneous
Thermal output	2 kW
Temperature	25°C
Fuel	Uranyl sulphate
Enrichment	20%
Reflector	Graphite
Max. neutron flux	
thermal	$6 \cdot 10^{10}$ n/cm^2 s
fast	$12 \cdot 10^{10}$ "
Experimental positions	1 thermal column
	1 through tube
	8 positions in reflector

Atmospheric Fluidized-bed Test Rig



Pressure	1 bar
Bed temperature	800-900°C
Thermal power	200 kW
Bed diameter	0.5 m
Fluidizing velocity	1-2 m/s

High-pressure Water Loop



Pressure	221 bar
Temperature	375°C
Flow	10 l/s
Test section	
Length	9 m
El-power	660 kW DC

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